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Physics Studies for Sodium Cooled ATW Blanket*

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ABSTRACT

Because the choice of blanket technologies is among the most important technical decisions faced in the Accelerator Transmutation of Waste (ATW) program, extensive system studies have been pursued on blanket design. A wide range of potential transmuter configurations and fuel cycle scenarios have been investigated using sodium, lead-bismuth eutectic (LBE), and gas as coolant. The primary objective has been to define the characteristics of a system that effectively consumes transuranics (TRU) separated from LWR spent fuel and minimizes TRU losses to the waste stream. For the liquid-metal cooled design studies, a fission-power level of 840 MWt was targeted, the same level previously adopted for the PRISM Advanced Liquid Metal Reactor (ALMR). The blanket is fueled with a non-uranium metallic dispersion fuel; pyrochemical techniques are used for recycle of residual TRU in this fuel after irradiation. Parametric studies were performed to optimize the sizing of the sodium cooled transmuter blanket, to mitigate the power peaking problems near the source region, and to assess startup core performance; results are summarized in this paper. Performance evaluations indicate that an average discharge burnup of 275 MWd/kg (29.5 atom%) is achieved with a 3.5 to 4 year fuel residence time. Reactivity loss over the half-year cycle is 4.9%Δk. The sodium coolant allows a compact core design with the volume reduced by ~1/3 compared to LBE cooled ATW blanket designs.

1. INTRODUCTION

Trade and system studies were initiated in the US in FY 2000 to evaluate the efficacy of various technical options for ATW system configuration.[1] The studies at ANL have so far focused primarily on the blanket component of the overall system, because the choice of blanket technologies is among the most important technical decisions faced in the ATW program. Both the basic technology and the particular features of the blanket design strongly impact transmutation performance and requirements on other ATW sub-systems (spallation target, accelerator, chemical separations). The sodium-cooled concept developed here is one of several blanket technology options currently under consideration in the ATW program. It is planned to conduct screening evaluations leading to the selection of two or three of the candidate concepts for further development, and later to select a single preferred technology from among those retained in the initial screening process.

A discussion of the issues associated with the use of alternative coolants for ATW can be found in the reports of the ATW roadmap working groups [2,3]. An in-depth summary of the key neutronic, thermal hydraulic, material compatibility, coolant chemistry, and coolant activation characteristics of various liquid metal coolants is provided in Reference 4. The sodium-cooled blanket design study builds on previous analyses conducted for an LBE cooled system [5]; similar design objectives and analysis techniques are utilized in this work.

In Section 2, the basic objectives and system assumptions for the sodium-cooled blanket design studies are described. The specific performance objectives and design constraints are identified in Section 3; the calculation methods are briefly described in Section 4. Parametric studies conducted to evaluate tradeoffs associated with adoption of various design parameters are presented in Section 5. The primary focus of these studies is development of a compact core and mitigation of power peaking problems inherent to the source driven configuration. In Section 6, the performance of startup and recycle fuel management scenarios is compared. Section 7 summarizes key conclusions.

2. OBJECTIVES AND SYSTEM ASSUMPTIONS

The primary objective of the system development efforts at ANL has been to achieve efficient transmutation of the transuranic actinides (TRU) separated from LWR spent fuel. It is generally recognized that a fast neutron energy spectrum is needed to accomplish the transmutation of minor actinides efficiently, because the fission-to-capture ratio for several key TRU nuclides is significantly greater in a fast spectrum [6]. The higher capture probability per incident neutron in a thermal spectrum causes build-up of the higher actinide fraction in the proportion of the TRU loading not consumed by fission, which adversely impacts neutron balance at high burnup and complicates recycle if the burnup is incomplete. On the other hand, the higher TRU inventory of fast systems for a given power level implies a lower specific power and a correspondingly lower burnup rate. Moreover, the fuel irradiation time in a fast spectrum is limited by radiation damage to structural materials caused by the large flux of high-energy neutrons. Consequently, fuel burnup in a fast system is generally incomplete in one pass through the transmutation blanket, and *recycle of discharged fuel is required to achieve an acceptably low TRU content in the waste stream.* It is assumed in the point design for the sodium cooled system that fuel recycle is performed using pyrochemical techniques referred to as "PYRO-B" in the ATW roadmap [1].

The major assumptions made in developing the sodium system point design are similar to those employed in the ATW roadmap as a basis for estimating ATW system costs and analyzing deployment scenarios; they can be summarized as follows:

1. A high-power linear accelerator generates a beam of energetic (~ 1 GeV) protons for delivery to target/blanket "transmuter" system; the proton beam impinges on a spallation target and produces a source of neutrons that drives the subcritical blanket.

2. Beam delivery to the target is in the vertical direction. Several target options are under consideration for use in conjunction with the sodium cooled blanket; these options include a liquid lead-bismuth eutectic (LBE) target, a sodium cooled tungsten target, and a gas cooled tungsten target. For this study, the LBE target is assumed.

3. The blanket is fueled with solid, uranium-free fuel clad with a low-swelling stainless steel alloy similar to the HT9 alloy developed in the U.S. Advanced Liquid Metal Reactor Program [7,8]. The fission power level of each transmuter module is 840 MWt -- consistent with the ALMR power level selected on the basis of favorable economics (through modular fabrication and installation) and excellent safety (passive removal of decay heat using ambient air as an inexhaustible heat sink).

4. Chemical separations required to extract uranium and fission products from the LWR discharge fuel are performed with the UREX process [1], and the TRU-containing output stream from this process is treated with a subsequent pyrochemical process "PYRO-A"

[1] to produce metallic TRU feed for use in ATW fuel fabrication. Recovery of the TRU remaining in the ATW fuel after irradiation in the ATW blanket is performed using the "PYRO-B" [1] process.

5. Key long-lived fission products (I-129 and Tc-99) are separately recovered during the LWR spent fuel pre-treatment steps. It has not yet been decided whether to immobilize these species in suitable waste forms or to transmute them in the ATW blanket; thus, initial system development efforts have focused on transmutation of TRU only.

The central objective of the system development studies conducted to date has been to define the characteristics of a transmutation system that minimizes transuranic losses to the waste streams. It should be emphasized that neither the feasibility nor optimality of the system developed with this objective has been demonstrated; for example, alternative systems may exhibit superior economic or safety performance.

As shown in Section 3, the objective of minimizing TRU losses to the waste stream is accomplished by maximizing the discharge burnup of ATW fuel (to minimize the number of recycle passes) and minimizing the fractional TRU loss per pass in recycle and refabrication. The achievable discharge burnup is likely constrained primarily by the fast-neutron irradiation damage to the cladding (fast fluence limit). The discharge burnup value currently targeted (~30 atom%) is high for conventional LMR fuels and remains to be demonstrated for the metallic dispersion fuel currently identified as the reference fuel. However, this burnup appears to be a reasonable development goal for the dispersion fuel type, particularly uranium-free fuels employing a non-fissioning matrix (e.g., zirconium or molybdenum); at a fixed heavy atom (fractional) burnup, the fission product density is much lower with a non-fissioning matrix than with a uranium matrix. Thus to the extent achievable fuel burnup is governed by fission product accumulation, higher burnup fractions can be targeted for non-uranium fuels.

3. PERFORMANCE OBJECTIVES AND CONSTRAINTS

In this study, full advantage is taken of the parametric studies performed as part of the lead bismuth eutectic (LBE) cooled ATW transmuter study [9]. In particular, the zirconium matrix metallic-dispersion fuel form developed for the LBE system is employed. In addition, the performance objectives reflect the general trends observed in the LBE cooled ATW trade studies (e.g., a preference for low inventory transmuter options).

The main purpose of the ATW system is to facilitate spent fuel disposal by removing the transuranic (TRU) elements and possibly long-lived fission products (LLFP) from the spent fuel and transmuting these constituents in the ATW blanket. Accordingly, the overriding performance objective for the ATW system is to minimize the fraction of the initial TRU inventory that is not transmuted and lost to the waste stream. Because there is a fraction of the TRU inventory lost every time the material is processed, developing high efficiency processing technology and limiting the number of processing operations required are design targets. From the viewpoint of transmuter design, the key design goal is to **maximize the discharge burnup**; this implies the fewest number of recycle/refabrication campaigns to destroy a given amount of material.

The primary design goal to eliminate TRU from the final waste stream is also the reason for utilizing **uranium-free fuel forms**. This prevents the generation of new TRU by in-reactor transmutation of uranium into plutonium. Because all current reactors operate on

uranium-based fuel forms, a fuel development program will be required for this waste transmutation mission; accommodation of high burnup is a primary development goal for this new fuel form. For this application, a dispersion fuel where TRU-10Zr fuel particles are dispersed in a zirconium metal matrix has been proposed [1]. Extensive experience with U-10Zr and U/Pu-10Zr fuels in the EBR-II fuel development program demonstrated the compatibility of similar fuel forms with sodium coolant; sodium was also utilized as a bond material within the pin. The dispersion fuel will be designed so that the fission products are contained within the fuel particles, which are contained within the matrix. Thus, this fuel form is expected to have superior irradiation performance (much less swelling than metal plutonium alloys) and there is no *conceptual* limit to the burnup.

Given the design goal of high discharge burnup, the required fuel irradiation time will be roughly proportional to the TRU inventory of the transmuter blanket. Low inventory options offer several advantages from a global fuel cycle perspective. Although the total amount of material destroyed is dictated solely by the power level, with low inventory of TRU a higher rate of burnup accumulation (MWd/kg per year or atom%/year) is achieved. In addition, low inventory transmuters require less of the TRU inventory targeted for transmutation to run a single system; thus, additional transmuters can be started from the same initial inventory, increasing the overall destruction rate. Low inventory systems also have a smaller final discharge (un-transmuted) inventory at the end of their operation campaign.

Conversely, design options with high inventory offer several improvements to the performance of the transmuter blanket. One of the major problems associated with utilization of uranium-free fuel is the complete loss of internal conversion of the uranium matrix, which creates fissile material (Pu-239) with fuel burnup. This leads to magnified reactivity losses with fuel burnup compared to conventional systems. For the "pure burner" design [6] developed for the ALMR for weapons plutonium disposition, reactivity loss rates were reduced by maximizing the TRU inventory; this was done by increasing the core volume and adding a fixed poison. Since the TRU loss rate is determined solely by the system power level, this effectively increases the ratio of end-of-cycle TRU mass to beginning-of-cycle TRU mass for a given cycle length, leading to smaller reactivity loss rates. In general, high inventory designs will be larger than low inventory options. The associated reduction of power density may be desirable to reduce the heat loads for the unproven non-uranium fuel forms. On the other hand, there will be economic penalties associated with any blanket size increase.

The relative performance of high and low inventory ATW design options was evaluated in Ref. 5 for the LBE-cooled ATW system. Results indicate that reactivity losses can be kept reasonably small through the use of sufficiently short cycle lengths (e.g., 3 to 6 months). The resulting reactivity losses can be compensated by a combination of increased source strength or potential reactivity insertion. Therefore, **low TRU inventory** has been adopted as a preferred option for the sodium-cooled transmuter studies. This approach will result in compact (low cost) design options with superior fuel cycle performance.

For conventional fast reactor systems, fuel pin integrity is ensured by imposing discharge burnup limits for the fuel matrix and peak fast fluence limits for the cladding material. Typically, the fuel lifetime is constrained to 4-5 years at which point the irradiated fuel is near both limits. For the zirconium matrix dispersion fuel employed here, there is no conceptual burnup limit. Therefore, the fuel lifetime will be constained only by damage considerations for the structural materials. For this analysis, a *peak fast fluence limit of* 4.0×10^{23} *n/cm*^{2} is assumed; this value is based on data for low-swelling ferritic alloy (HT-9) developed in the ALMR program [7,8].

The TRU fraction in the fuel is determined such that the *effective multiplication factor at the beginning of cycle is 0.97.* The reactivity loss and declining source multiplication during the operating cycle are assumed in the current design analyses to be compensated by increased source strength. To prevent a need for accelerator capacity to increase by more than a factor of three, the *end of cycle effective multiplication factor must be at least 0.91*; this limit effectively constrains the cycle length for a given reactivity loss rate.

A *maximum volume fraction of 50% TRU-Zr fuel particles* in the proposed dispersion fuel is assumed. However, the dispersion fuel will be easier to fabricate and likely have improved irradiation performance at much lower fuel volume fractions. Thus, design options that result in reduced fuel volume fraction within the dispersion matrix are preferred.

Finally, the power density of the transmuter blanket is constrained by fuel heat load and heat transfer considerations. In particular, the peak linear power is constrained by the need to limit peak fuel centerline temperatures to prevent fuel melting. Estimates of the peak linear power limit for the proposed fuel form and for ternary (U-Pu-10Zr) metallic alloy fuel, which was the reference fuel form in the PRISM ALMR [7], are derived in Table 1. The TRU-10Zr composition of the fuel particles in the ATW dispersion fuel has a significantly lower (~150°C) solidus temperature compared to the ternary metal fuel alloy. However, the thermal conductivity of the dispersion (composite) fuel is significantly improved because the zirconium matrix is highly conductive and its thermal properties are not expected to degrade with irradiation as observed for fuel alloys. The net result is an estimated increase in the allowable peak linear power from 375 W/cm to 450 W/cm. Note that the peak linear power limit is lower with LBE coolant because of a much larger temperature rise in the coolant itself. This difference is attributed primarily to higher coolant flow rate and improved thermal conductivity in the sodium. Given the large uncertainties associated with thermal properties of the non-uranium dispersion fuel, it was considered prudent to impose a conservative design constraint. Thus, a *peak linear power limit of 400 W/cm* was assumed in these parametric studies.

4. COMPUTATIONAL METHODS AND TECHNIQUES

Analyses of the sodium-cooled system point design have so far focused primarily on the equilibrium fuel cycle, because system performance under equilibrium conditions is believed to be a good basis for design optimization (startup cycle performance is compared to the equilibrium results in Section 6). Equilibrium cycle performance characteristics were calculated using the REBUS-3 fuel cycle analysis code [10,11]. The region-dependent multigroup cross sections used in the neutronic analyses were originally generated for the ALMR pure burner design as described in Ref. 7; they are based on ENDF/B-V.2 basic data processed using the MC^2-2 [12] and SDX [13] codes for a 21-group energy structure.

In the equilibrium fuel cycle model, the charged fuel contains the transuranics recovered via recycle from the discharged ATW fuel, supplemented by LWR-discharge TRU to make up for the TRU consumed by fission. Determination of the equilibrium composition neglected the very small proportion of TRU lost during recycle and refabrication, and assumed 5% of rare-earth fission products carried over by the recycled ATW TRU. The TRU mass loading in the fuel which meets the targeted subcriticality level at BOEC was determined using the REBUS-3 enrichment search techniques. REBUS-3 also computes both batch-dependent and batch-averaged compositions at BOEC and EOEC for each specified depletion region. In this study, five (equal length) axial depletion zones were consistently used; in the planar dimension, depletion zones consisted of individual fuel assemblies or groups of neighboring assemblies with similar reaction rates. Irradiation swelling of the fuel was modeled in the depletion

Table 1. Estimation of Peak Linear Power for ATW Dispersion Fuel (U-10Zr/Zr) and ALMR Metallic Alloy Fuel (U-27Pu-10Zr)

Parameter	ALMR Metal Fuel Sodium Coolant	ATW Dispersion Fuel Sodium Coolant	ATW Dispersion Fuel LBE Coolant
Coolant Film Heat Transfer Coefficient, W/m^2K	1.42E5	1.42E5	3.77E4
Clad Thermal Conductivity, W/mK	26.8	26.8	26.8
Fuel Solidus Temperature, °C	990	840	840
Fuel Thermal Conductivity, W/mK	15.3	13.8	13.8
Matrix Thermal Conductivity, W/mK	-	22.2	22.2
Fuel Volume Fraction, %	100	~40	~30
Irradiation Decrease in Fuel Conductivity, %	50	50	50
Effective Fuel Matrix Thermal Conductivity, W/mK	7.65	16.1	17.6
Peak Linear Power Estimate, W/cm	**374**	**454**	**332**

calculations as a uniform 5% axial expansion of the fresh fuel, based on IFR experiments for U-Pu-Zr ternary metal fuel, even though the proposed dispersion fuel is expected to exhibit less irradiation swelling.

Preliminary sensitivity studies of the effect of various flux computational options available in REBUS-3 were performed as part of the LBE cooled ATW blanket design studies [9]. Both the inhomogeneous source problem and the corresponding homogeneous eigenvalue problem (i.e., a system without the spallation source made artificially critical by use of an eigenvalue to scale neutron production) were considered. These sensitivity studies demonstrated that the global performance parameters are very similar for different flux calculation methods. In addition, the integral parameters estimated with the eigenvalue calculations were found to agree well with the results of the corresponding inhomogeneous source calculations; peak flux and power were not as accurately predicted by the eigenvalue calculations. Thus, for computational convenience, homogeneous (eigenvalue) neutronic calculations performed using the hexagonal-Z nodal diffusion option of DIF3D were employed as a basis for optimizing the global design parameters (e.g., system size) of the sodium-cooled ATW blanket as described in Section 5.1.

The increase in source strength required to compensate the lower EOEC neutron multiplication leads to large increases in flux in the vicinity of the source region, which creates flux and power peaking problems. In Section 5.2, design options to mitigate the power peaking inherent to the source-driven configuration are investigated. For this analysis, inhomogeneous source problems were solved using a "generic" spallation neutron source distribution gener-

ated for a 1 GeV proton beam and a prototypic LBE target [14]. Even though the spallation neutron source distributions need to be generated for specific transmuter (target/blanket) configurations, inhomogeneous source analyses can be performed with sufficient accuracy using generic source distributions appropriate to the accelerator beam proton energy and the spallation target material and geometry. For these analyses, the flux calculation method was switched to the triangular-Z finite difference option of DIF3D [15] to estimate the peak values more accurately.

5. PARAMETRIC DESIGN STUDIES

5.1. Core Sizing Study

A preliminary 840 MW(thermal) LBE cooled ATW transmuter design was used as the starting point for developing the sodium cooled system. This LBE configuration employs the seven central assembly locations for the LBE target and buffer as shown in Fig. 1. The preliminary LBE-cooled blanket is composed of 192 fuel assemblies in two "enrichment" zones. The enrichment is varied by using a higher volume fraction of the TRU-10Zr fuel particles within the zirconium matrix in the outer region; an enrichment split of 1.2 was assumed.

For the LBE coolant, the lattice design is quite loose with a pitch-to-diameter (P/D) ratio of 1.73 resulting in a coolant volume fraction of nearly 70%. This high coolant volume fraction is necessary to achieve the low coolant velocity required for the heavy liquid metal coolant. In addition, several dummy structural pins were employed for holding down the assembly; this hold-down mechanism is required with the heavy liquid metal coolant but not with sodium coolant. The net result is a low fuel volume fraction of only 20% as compared to a smeared fuel volume fraction of 38% in the ALMR design [7].

First, the performance effects were evaluated for a direct replacement simple exchange of LBE coolant with sodium, using the loose (LBE) lattice design. In addition, the fuel cycle

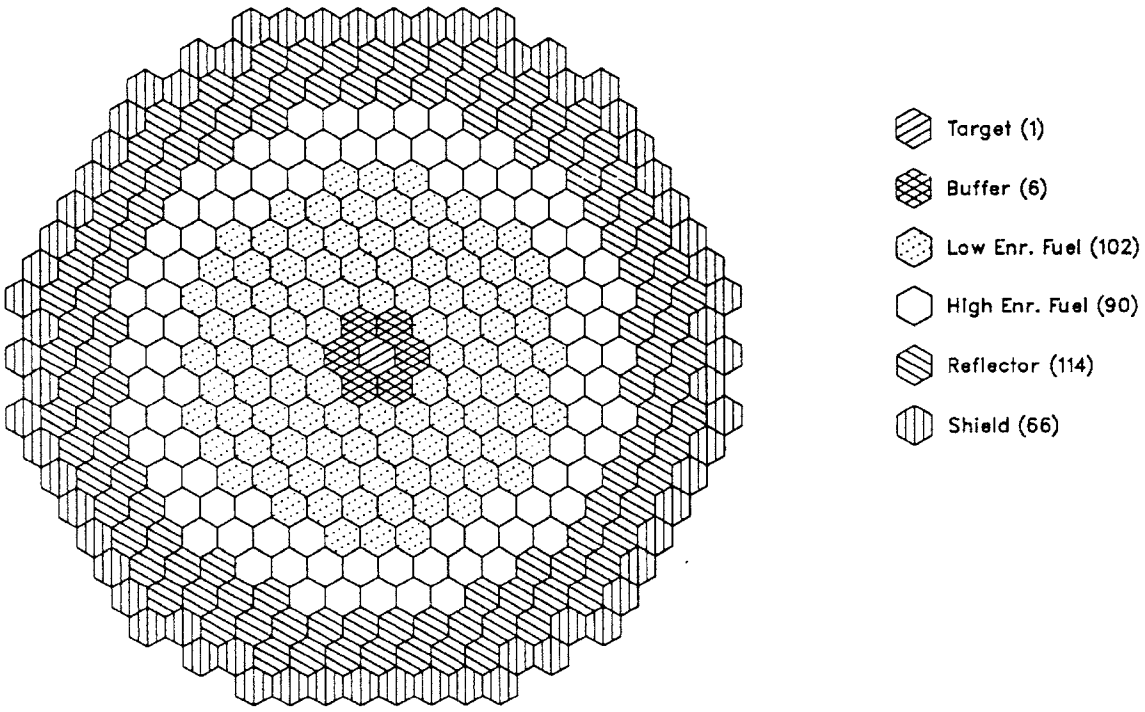

FIG 1. LBE Cooled ATW Blanket Configuration

parameters of the LBE system were retained, i.e., a fuel residence time of six 145 day cycles was assumed. This evaluation illustrates the impact of sodium coolant (relative to LBE) on the reactor performance. Performance results are compared for the two coolants in Table 2. From a reactor physics viewpoint, the main difference between the sodium and LBE coolant is increased scattering (without moderation) in the LBE. Thus, the LBE reduces neutron escape from the interior regions of the blanket and provides superior reflection for neutrons that leak out of the active zone. Thus, a much higher (~30%) TRU inventory is required to achieve the beginning-of-equilibrium-cycle (BOEC) multiplication target of 0.97 when sodium coolant is used. For a fixed fuel lifetime, a corresponding decrease in the average discharge burnup (by 20%) is observed. As identified in Section 3, the high inventory has a favorable impact of reducing the reactivity loss rate. In addition, the power peaking factors are lower in the sodium system because the LBE coolant retains more neutrons in the peak power regions.

The low fast fluence for the sodium case (2.5 vs. 4.0×10^{23} n/cm^2 for the LBE case) indicates that fuel lifetime can be extended. A scoping study indicated that the fuel lifetime can be extended from 6x145 = 870 effective full-power days (efpd) to 1400 efpd before the fluence limit is exceeded for the sodium cooled case. Results calculated for a fuel management scheme of eight 175 day irradiation intervals (1400 efpd fuel lifetime) are also given in Table 2. A corresponding increase in average burnup from 218 MWd/kg to 314 MWd/kg is observed. Because the *average* burnup of the blanket also increases, a 4% increase in the TRU inventory is required to maintain the BOEC multiplication factor. An important result is that *the sodium-cooled design achieves a higher average discharge burnup than the LBE-cooled design* (314 MWd/kg vs. 272 MWd/kg) *at the same discharge fluence level.* This difference is attributed to the moderating effect of the sodium coolant. The neutron energy spectrum is harder when the LBE coolant is utilized, resulting in a higher fast fluence to total flux ratio. This difference is particularly pronounced for lattice designs such as the LBE cooled ATW configuration where the coolant volume fraction is quite high.

Table 2. Performance Parameters of LBE Configuration with LBE and Sodium Coolant

LBE Configuration and Assembly Design		LBE-Cooled	Sodium Substitution	Extended Fuel Lifetime
Number of fuel batches		6	6	8
Cycle irradiation time (days)		145	145	175
Fuel particle fraction (volume % in matrix)	Inner zone	27.5	34.2	38.2
	Outer zone	33.5	41.8	**46.8**
Multiplication Factor	BOEC	0.970	0.970	0.969
	EOEC	0.912	0.928	0.922
Burnup reactivity loss (% Δk)		5.80	**4.17**	4.69
Peak linear power (W/cm)		317	287	311
Discharge burnup (MWd/kgHM)		272	**218**	**314**
Peak fast fluence (10^{23} n/cm^2)		3.96	2.55	4.03
BOEC Heavy metal inventory (kg)		2256	**2899**	3024

One drawback of the design where sodium is simply substituted in the LBE configuration is that the volume fraction of fuel particles (47% in the outer zone) is close to the 50% limit. Moreover, the resulting configuration does not take advantage of the possibility of designing a more compact system (reduced coolant volume fraction at higher flow rate) with sodium coolant. Thus, modifications to the assembly design which increase the fuel volume fraction were investigated; these changes effectively allocate additional space for zirconium matrix material. The most significant change was to switch from the loose lattice (P/D ~1.7) LBE design to a conventional tight lattice (P/D ~1.2) design. The assembly design proposed for the sodium-cooled ALMR [7] was utilized. This change nearly doubles the fuel matrix volume as the smeared fuel volume fraction increases from 20% to 38. Furthermore, the smear density of the fuel was increased from 75% to 85%. Significant swelling of the dispersion fuel is not expected; thus, the large gaps required to accommodate fuel swelling in the ALMR ternary metal fuel design can be reduced.

Using this assembly design, the fuel particle fraction decreased to ~20%, roughly half the volume fraction required for the loose lattice assembly design. The exchange of coolant for fuel matrix material and additional structural material lead to a 10% increase in the required TRU inventory as a result of increased leakage and parasitic capture in the structural materials. The higher inventory decreases the average discharge burnup to 285 MWd/kg. However, the fast fluence did *not* decrease despite the reduction in discharge burnup. This is attributed to spectral hardening associated with the decreased coolant volume fraction in the tight lattice design. The tight lattice sodium design exhibited performance characteristics similar to the LBE cooled configuration with a 50% higher TRU inventory for the same net TRU consumption rate. The specific TRU consumption rate of the LBE design was 10.5% per year as compared to 7% per year for this sodium-cooled option. Therefore, design options to reduce the TRU inventory of the sodium-cooled design were explored next.

The ALMR tight lattice design introduces an additional row of pins within an assembly of the same physical dimensions. This reduces the peak linear power from 311 W/cm to 260 W/cm despite larger power peaking factors in the tight lattice case. There remains significant margin to the peak linear power limit of 400 W/cm derived in Section 3. Thus, more compact configurations using the tight lattice design were evaluated. In addition to lowering the TRU inventory requirements, reductions in the core volume provide the economic benefit of reducing the system size and blanket hardware requirements (e.g., fewer fuel assemblies).

For this evaluation, the core height and assembly design were retained, and fuel assemblies were progressively eliminated from the periphery of the ATW configuration. Results are given in Table 3 for cases where the number of fuel assemblies was reduced from 192 to 132 and 120 assemblies. To avoid exceeding the assumed fluence limit, the fuel residence time must be decreased roughly in proportion to the volume decrease. For both revised configurations, a six cycle management scheme was utilized with the 175 day cycle length retained. The 120 assembly case has a peak linear power of 419 W/cm which exceeds the design limit as well as a peak fast fluence of 4.45×10^{23} n/cm^2. Thus, the 132 assembly configuration shown in Fig. 2 was identified as a favorable size for the sodium-cooled blanket configuration. The number of fuel assemblies is reduced by ~30% compared to the LBE cooled blanket. The specific consumption rate increases to 9.5% per year. The 372 W/cm peak linear power and 3.83×10^{23} n/cm^2 leave some margin to the design limits to accommodate increased flux and power peaking when the inhomogeneous source is modeled as evaluated in Section 5.2.

Table 3. Performance Parameters of Compact Sodium Cooled Configurations

		192 Fuel Assemblies	132 Fuel Assemblies	120 Fuel Assemblies
Number of fuel batches		8	6	6
Cycle irradiation time (days)		175	175	175
Fuel particle fraction (volume % in matrix)	Inner zone	19.6	21.6	21.4
	Outer zone	23.7	26.2	25.9
Multiplication Factor	BOEC	0.970	0.970	0.970
	EOEC	0.912	0.913	0.905
Burnup reactivity loss (% Δk)		4.51	5.68	6.48
Peak linear power (W/cm)		260	372	**419**
Discharge burnup (MWd/kgHM)		286	281	310
Peak fast fluence (10^{23} n/cm^2)		4.00	3.83	4.45
BOEC Heavy metal inventory (kg)		3373	**2602**	2320

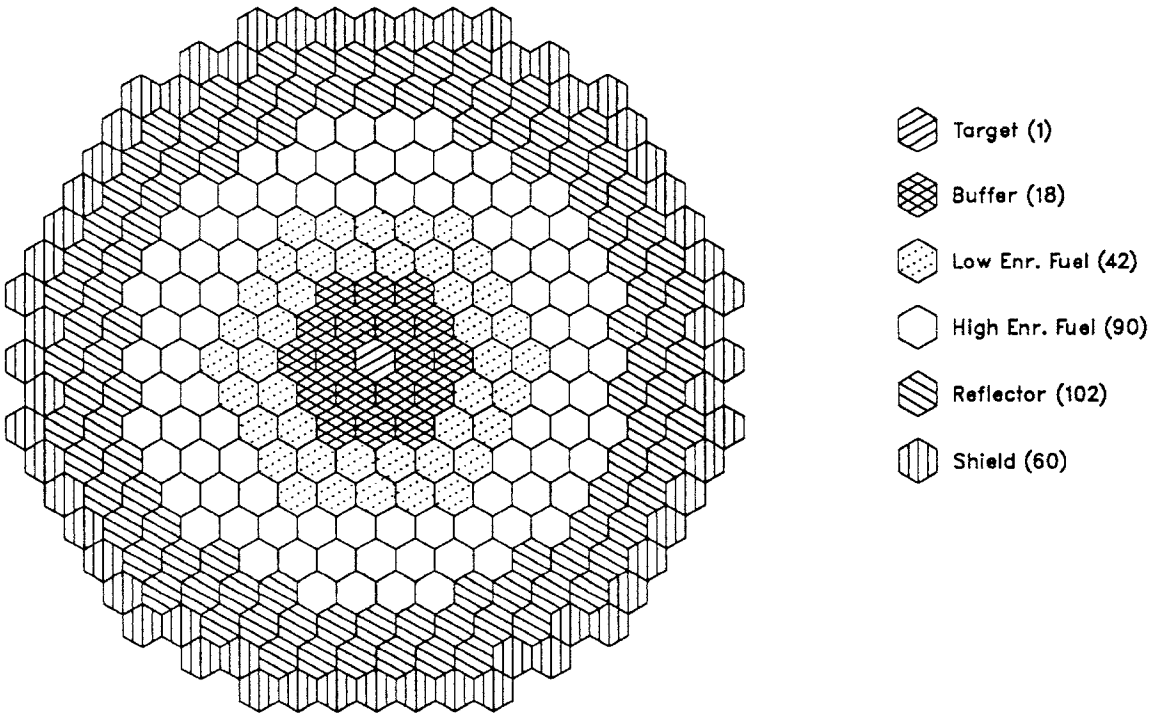

FIG. 2. Sodium-Cooled ATW Blanket Configuration

5.2. Power Peaking Study

In this section, design options to enhance the flux and power peaking performance of the sodium-cooled ATW blanket are evaluated. For this investigation, it is important to consider the impact of the inhomogeneous source on the flux and power distributions. Performance results for the eigenvalue neutronics computation and the inhomogeneous source model are compared in Table 4. The fuel enrichment requirements and mass flows are very close; the TRU inventory is ~1% greater for the source-driven computation. These results confirm that the eigenvalue calculation adequately predicts the global performance parameters for the modest subcriticality levels of interest. However, significant differences are observed in the EOEC power peaking. Because the multiplication factor is lower at EOEC, the required neutron source strength increases by roughly a factor of three (if no other reactivity control techniques are used) to maintain the power output. This leads to a large flux peak in the interior fuel assemblies close to the spallation target. This phenomenon can also lead to a power peak in the innermost row of assemblies (increases from 372 to 459 W/cm). In addition, the fast fluence in this innermost row increases to 4.83×10^{23} n/cm^2.

Several design options can be conceived to mitigate this peaking behavior:

- The enrichment zoning of the blanket can be tailored to suppress the peak.
- The cycle length can be shortened to reduce the decline in multiplication factor between BOEC and EOEC.
- The blanket size can be increased to reduce the power density and specific power; the reduced cycle burnup also mitigates the multiplication factor decline.
- Fuel shuffling can be employed to preferentially place high burnup assemblies and limit exposure time in the inner locations of the blanket near the source
- Fewer irradiation cycles can be employed for the innermost fuel assemblies. This does not reduce the flux or power peaking itself but does reduce the discharge fast fluence for the limiting (interior) fuel assemblies.

Increased blanket size is an undesirable option because the TRU inventory would increase as shown in the previous subsection. Fuel shuffling could be utilized to accommodate the power peak, but in-residence fuel movement complicates fuel handling. The final option (decreased inner region residence) is also not favored because it penalizes the discharge burnup of the inner region fuel. Thus, the current parametric studies have focused on the first two options (enrichment zoning and reduction of cycle duration). Furthermore, a tentative limit of two enrichment zones was imposed for this study. Additional enrichment zones could be employed to better flatten the power shape, but utilization of numerous enrichments (different fuel particle volume fractions within the dispersion matrix) complicates both fuel fabrication and fuel handling.

First, alternate allocations of the assemblies to high and low enrichment zones were investigated with the inner-to-outer blanket zone enrichment split of 1.2 retained. The most favorable performance was observed when the low enrichment zone was sized at two rows thickness; this results in only 42 low enrichment assemblies, as shown in Fig. 2. The large number of high enrichment assemblies leads to a lower power level in the low enrichment zone with a compensating power increase in the high enrichment. The power peaking factor decreases from 1.71 to 1.625; however, the *flux* does not decrease in the interior region because the EOEC multiplication factor is not significantly affected; thus, the peak fast fluence is not significantly reduced.

Table 4. Selected Power Peaking Study Performance Results

Type of Computation		Evalue	Source	Source	Source
# of High Enrichment Assemblies		66	66	42	42
Number of Fuel Batches		6	6	**8**	8
Cycle Irradiation Time		175	175	**135**	135
Enrichment Split		1.2	1.2	1.2	**1.3**
Fuel particle fraction (volume % in matrix)	Inner zone	21.6	21.7	21.2	20.2
	Outer zone	26.2	26.3	25.7	26.6
BOEC Heavy metal inventory (kg)		2602	2623	2617	2638
Multiplication Factor	BOEC	0.970	0.970	0.970	0.971
	EOEC	0.913	0.907	**0.921**	0.921
Burnup reactivity loss (% Δk)		5.68	6.31	4.92	4.95
Power peaking factor	BOEC	1.467	1.470	1.478	1.508
	EOEC	1.447	**1.708**	1.566	**1.515**
Peak linear power (W/cm)		372	**459**	427	**399**
Discharge burnup (MWd/kgHM)		281	280	285	283
Peak fast fluence (10^{23} n/cm^2)		3.83	**4.83**	4.71	4.55

To reduce flux peaking, the EOEC multiplication factor can be increased by using a fuel management strategy with a shorter cycle length. Thus, conversion to an eight batch scheme with the cycle length reduced from 175 days to 135 days (roughly conserving fuel lifetime) was evaluated; results are given in Table 4. The EOEC multiplication factor increases from 0.906 to 0.921; this implies a ~20% decrease in source intensity. Since the source is centrally located, this further reduces the inner row peaking factor from 1.71 to 1.57 with a peak linear power of 427 W/cm (close to the 400 W/cm limit). The reduced source strength also decreases the flux in the interior assemblies; thus, a slight decrease in the peak fast fluence is observed in this case.

Finally, variations in enrichment split between the high and low enrichment zones were evaluated. Using the blanket configuration shown in Fig. 2, the enrichment split was varied between 1.2 and 1.8. As the enrichment split is increased, the BOEC power peaking factor increases because the power peak is located in the outer core. Conversely, the EOEC power peak initially decreases because it is located in the inner core region. At an enrichment split of roughly 1.3 the peak EOEC power in the inner and outer zones are roughly equal, and the lowest peaking factor of 1.515 is obtained. At higher enrichment splits, the EOEC power peak is located in the outer core, thus peaking becomes more severe with increasing outer zone enrichment. Parametric results show that the peak fast fluence also decreases with increasing enrichment split. The fluence peak is located in the inner (low enrichment) zone, and shifting of the TRU loading (and fission rate) into the outer region reduces the inner zone flux. Based

on these results, an enrichment split of 1.3 was specified for the sodium-cooled system point design. This is the only split which meets the peak linear power limit of 400 W/cm. Since the peak discharge fast fluence exceeds the assumed limit for all cases, a modified fuel cycle is required. In the final recommended point design, the fuel lifetime for the inner (low enrichment) fuel assemblies is reduced from eight to seven cycles; this results in discharge fast fluence within the design limit with only a slight penalty in the TRU burnup performance.

6. COMPARISON OF STARTUP AND RECYCLE SCENARIOS

In this section, the performance of the sodium-cooled ATW design operating on a startup cycle (i.e., using LWR-discharge transuranics for the ATW feed stream) is contrasted to the base equilibrium (with recycle) case. For the startup case, an equilibrium REBUS-3 calculation is performed with processed LWR transuranics as the sole source of fuel material (no recycled feed). This computation roughly models the behavior of the ATW blanket in its initial core loadings and the condition of the fuel material for its first pass through the transmutation system. The comparison of startup and equilibrium cycle performance was performed for the blanket configuration developed in Section 5 (Fig. 2) with an enrichment split of 1.3. The cycle length of 135 days (~1/2 year at a 75% capacity factor) was retained. An eight batch fuel management strategy was employed for the high enrichment fuel assemblies and a seven batch strategy for the interior low enrichment fuel assemblies.

Performance results for the startup case are compared to the recycle case in Table 5. The TRU volume fraction and TRU inventory for the startup case are ~20% lower than the equilibrium recycle; this is attributed to changes in the TRU isotopics as discussed below. The reduced inventory leads to a greater reactivity loss over the burnup, which in turn exacerbates the EOEC power peak. It appears a higher enrichment split and/or shorter cycle length are desirable to reduce the power peaking in the initial (low inventory) loadings. On the positive side, the reduced inventory yields a proportional increase in the average discharge burnup with associated fuel cycle performance benefits.

Table 5. Performance Characteristics for Startup and Recycle Scenarios

		Recycle	Startup
Fuel particle fraction (volume % in matrix)	Inner zone	19.9	**16.0**
	Outer zone	26.2	**21.0**
Multiplication Factor	BOEC	0.970	0.971
	EOEC	0.920	0.909
Burnup reactivity loss (% Δk)		4.94	6.13
Power peaking factor	BOEC	1.501	1.453
	EOEC	1.508	**1.559**
Peak linear power (W/cm)		397	449
Discharge burnup (MWd/kg)		275	**340**
Peak fast fluence (10^{23} n/cm^2)		4.06	4.26
BOEC Heavy metal inventory (kg)		2620	**2025**

The evolution of the TRU isotopics in the ATW fuel cycle is illustrated in Table 6 where the charge and discharge compositions for the startup and recycle cases are compared. After the initial in-core residence, the Pu-239 fraction has decreased from 53% to 34%. The proportion of Pu-239 and other fissile nuclides is reduced relative to the fertile transuranics which tend to concentrate due to their lower cross sections. This phenomenon is the cause of the lower TRU enrichment requirements for the startup core where the fissile fraction is highest. The isotopics change significantly during the first irradiation campaign. The Pu-240 has already increased to nearly its equilibrium level (~33%). It takes longer for the higher capture products Pu-242, Am-243, and Cm-244 to reach their equilibrium concentration. The Am-241 fraction actually decreases because the initial feed has a much longer post-irradiation cooling time, yielding additional Pu-241 decay, than the ATW discharge and recycle compositions. The evolution of isotopic fractions displayed in Table 6 suggests a fairly rapid and smooth transition from the startup cycle performance to the equilibrium recycle performance.

Table 6. Evolution of TRU Isotopics (weight %) in the ATW Fuel Cycle

Isotope	Startup Cycle		Equilibrium Recycle		
	Initial Feed (LWR TRU)[a]	Once-Through Discharge	Equilibrium Feed[b]	Equilibrium Discharge	0.8y Cooled Eq. Discharge
U-234	0.000	0.080	0.468	0.580	0.621
U-235	0.004	0.012	0.110	0.153	0.154
U-236	0.002	0.013	0.149	0.204	0.207
U-238	0.478	0.642	1.022	1.249	1.249
Np-237	5.023	3.541	2.896	1.990	1.997
Pu-238	1.272	5.773	5.039	6.226	6.552
Pu-239	**53.196**	**34.254**	28.729	18.499	18.502
Pu-240	21.533	31.800	31.492	35.437	35.548
Pu-241	3.782	5.683	5.523	6.780	6.525
Pu-242	4.686	**7.285**	**10.555**	13.005	13.007
Am-241	8.967	6.831	6.850	5.068	5.316
Am-242m	0.014	0.565	0.340	0.480	0.478
Am-243	0.926	**1.800**	**3.404**	4.440	4.440
Cm-242	0.000	0.771	0.030	0.519	0.147
Cm-243	0.002	0.066	0.039	0.057	0.056
Cm-244	0.104	**0.763**	**2.471**	3.682	3.570

[a]Processed transuranics from medium burnup PWR at 25 years cooling [16].
[b]Equilibrium feed is a mixture of recycled ATW transuranics and processed LWR transuranics as required for makeup.

7. CONCLUSIONS

Parametric studies have been performed to optimize the sizing of the sodium cooled transmuter blanket, to mitigate power peaking problems near the source regions, and to assess startup core performance. In these studies, a wide range of potential transmuter configurations and fuel cycle scenarios were investigated for an assumed fission-power level of 840 MWt, typical size of modular fast reactor designs such as the ALMR.

Compared to ATW systems employing lead-bismuth eutectic (LBE) coolant, sodium cooled blankets require a higher TRU inventory because of increased neutron leakage. However, much higher flow rates can be used with sodium coolant which allows a significant reduction in the coolant volume fraction; for this study, the ALMR tight lattice (pitch-to-diameter ratio of ~1.2) fuel assembly design was employed. The associated increase in fuel volume fraction allows considerable compaction of the blanket (~30% compared to the LBE cooled design) with associated economic benefits. The extent of this size reduction is constrained by the peak linear power limit that was estimated to be 400 W/cm for the non-uranium metallic dispersion fuel in a sodium cooled environment.

Design options to enhance the flux and power peaking performance of the sodium cooled ATW blanket were also investigated. The increased source strength at end-of-cycle (EOC) can lead to severe flux and power peaks in the blanket near the source. Refined allocations of the assemblies to high and low enrichment zones were developed to reduce the power peaking factors. Parametric studies indicate that an enrichment split of 1.3 gives the most favorable performance. In addition, the cycle length was shortened to 135 days (half a year at 75% capacity factor) to reduce the decline in multiplication factor. Even with improved power peaking behavior, high flux levels in the inner blanket require a somewhat shorter fuel lifetime (compared to the outer blanket) for the same discharge fast fluence level.

The performance of the sodium cooled ATW blanket system point design operating on a startup cycle was contrasted to the equilibrium cycle results. The main performance difference is that the transuranic inventory is ~20% lower for the startup cycle because of the higher fissile content of the LWR discharge feed. The evolution of the transuranic isotopics suggests a fairly rapid and smooth transition from the startup cycle performance to the equilibrium cycle performance.

If successfully developed, the proposed sodium cooled system would consume LWR-discharge TRU at the maximum rate achievable per unit of fission energy produced (~0.9 g/MWtd). The overriding design objective of high discharge burnup was shown to be achievable in a configuration with high power density (enabling small system size and potentially favorable economics) and relatively low burnup reactivity loss (to reduce requirements for reactivity and/or source control). System design and operating characteristics that satisfy these goals while meeting key thermal-hydraulic and materials-related design constraints were preliminarily developed.

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